

Mr. James A. Gresham, Manager
Regulatory and Licensing Engineering
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

June 2, 2005

SUBJECT: DRAFT SAFETY EVALUATION FOR TOPICAL REPORT WCAP-15836-P,
"FUEL ROD DESIGN METHODS FOR BOILING WATER REACTORS -
SUPPLEMENT 1" (TAC NO. MB5740)

Dear Mr. Gresham:

By letter dated June 25, 2002, as supplemented by letters dated April 16, and July 30, 2004, March 9, and April 22, 2005, Westinghouse Electric Company (Westinghouse) requested review and approval of topical report (TR) WCAP-15836-P, "Fuel Rod Design Methods for Boiling Water Reactors - Supplement 1." Enclosed for Westinghouse review and comment is a copy of the staff's draft safety evaluation (SE) for the TR.

Pursuant to 10 CFR 2.390, we have determined that the enclosed draft SE does not contain proprietary information. However, we will delay placing the draft SE in the public document room for a period of ten working days from the date of this letter to provide you with the opportunity to comment on the proprietary aspects. If you believe that any information in the enclosure is proprietary, please identify such information line-by-line and define the basis pursuant to the criteria of 10 CFR 2.390. We intend to treat the entire Technical Evaluation Report as proprietary and not make it publicly available. After ten working days, the draft SE will be made publicly available, and an additional ten working days are provided to you to comment on any factual errors or clarity concerns contained in the SE. The final SE will be issued after making any necessary changes and will be made publicly available. The staff's disposition of your comments on the draft SE will be discussed in the final SE.

To facilitate the NRC staff's review of your comments, please provide a marked-up copy of the draft SE showing proposed changes and provide a summary table of the proposed changes.

If you have any questions, please contact Brian Benney at (301) 415-3764.

Sincerely,

/RA/
Robert A. Gramm, Chief, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 700

Enclosures: 1. Draft Safety Evaluation
2. Technical Evaluation Report (**Proprietary**)

cc w/encls: See next page

Westinghouse Electric Company

Project No. 700

cc:

Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

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DRAFT SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT WCAP-15836-P, "FUEL ROD DESIGN METHODS

FOR BOILING WATER REACTORS - SUPPLEMENT 1"

WESTINGHOUSE ELECTRIC COMPANY LLC

1.0 INTRODUCTION

1 By letter dated June 25, 2002 (Reference 1), as supplemented by letters dated April 16, 2004
2 (Reference 2), July 30, 2004 (Reference 3), March 9, 2005 (Reference 4), and April 22, 2005
3 (Reference 5), Westinghouse Electric Company (Westinghouse) requested review and
4 approval of WCAP-15836-P, entitled, "Fuel Rod Design Methods for Boiling Water Reactors -
5 Supplement 1." This licensing topical report (TR) describes improvements to the previously
6 approved Boiling Water Reactor (BWR) fuel performance codes STAV, VIK, and COLLAPS
7 (Reference 6). The new code versions, STAV7.2, VIK-3, and COLLAPS II, Version 3.3D, are
8 intended to support fuel design and licensing applications up to a rod average burnup of
9 62 GWd/MTU.

10
11 The Nuclear Regulatory Commission (NRC) staff's review was assisted by Pacific Northwest
12 National Laboratory (PNNL). The NRC staff's conclusions on the acceptability of WCAP-
13 15836-P are supported by the proprietary PNNL Technical Evaluation Report (TER) which is
14 being withheld from public availability.

15
16 2.0 REGULATORY EVALUATION

17
18 Regulatory guidance for the review of fuel system designs and adherence to applicable
19 General Design Criteria (GDC) is provided in NUREG-0800, "Standard Review Plan for the
20 Review of Safety Analysis Reports for Nuclear Power Plants" (SRP), Section 4.2, "Fuel System
21 Design." In accordance with SRP Section 4.2, the objectives of the fuel system safety review
22 are to provide assurance that:

- 23
24 a. The fuel system is not damaged as a result of normal operation and anticipated
25 operational occurrences (AOOs),
26 b. Fuel system damage is never so severe as to prevent control rod insertion when it
27 is required,
28 c. The number of fuel rod failures is not underestimated for postulated accidents,
29 and
30 d. Coolability is always maintained.

31
32 In addition to licensed reload methodologies, fuel performance models are utilized, along with
33 an approved mechanical design methodology, to demonstrate compliance to SRP Section 4.2
34 fuel design criteria. WCAP-15836-P describes improvements to Westinghouse's suite of BWR
35 fuel performance computer models (STAV7.2, VIK-3, and COLLAPS II, Version 3.3D). The

1 NRC staff's review of WCAP-15836-P is to ensure that these computer models are capable of
2 accurately (or conservatively) predicting the in-reactor performance of fuel rods and to identify
3 any limitations on the ability of the code to perform this task. A subsequent TR on fuel
4 mechanical design methodology (WCAP-15942-P) will discuss how a Westinghouse BWR fuel
5 design, employing these models, demonstrates compliance to the applicable regulatory
6 requirements identified in SRP Section 4.2.

7 8 3.0 TECHNICAL EVALUATION

9
10 The NRC staff's review of these fuel performance models is summarized below:

- 11 • Verify material properties based on supporting mechanical testing database.
- 12 • Verify each model (e.g., fuel temperature, creep, etc.) based on separate effects testing
13 and measurements.
- 14 • Verify synergistic interaction of coupled models based on comparisons to instrumented
15 in-pile test programs.
- 16 • Verify predicted in-reactor performance based on pool-side and hot-cell irradiation
17 database.

18
19
20 In addition to comparing the computer model predictions to the supporting database, PNNL
21 performed extensive bench marking of STAV7.2 against the NRC audit code FRAPCON-3.
22 The fuel performance models in FRAPCON-3 have been validated against an extensive
23 database and are continually assessed against newer data as it becomes available.

24 25 3.1 Thermal Modeling

26 27 Pellet Heat Generation and Heat Transfer Methods

28 The solution method for the heat generation within the pellet has been improved in STAV7.2.
29 Fuel and cladding temperatures are calculated assuming steady-state, radial-only heat transfer
30 from the pellet, across the pellet-cladding gap, through the cladding base metal, across the
31 oxide and crud layers, and across the water film to the coolant. The PNNL technical
32 assessment of the heat generation and heat transfer solution methods is provided in
33 Section 2.1 of the supporting TER. Based upon this assessment, the NRC staff finds the pellet
34 heat generation and heat transfer solution methods in STAV7.2 acceptable.

35 36 Fuel Thermal Conductivity

37 Based upon FRAPCON-3 benchmarks and comparisons to relevant empirical data, PNNL
38 concluded that the STAV7.2 UO₂ pellet thermal conductivity model was non-conservative
39 (Section 2.2 of supporting TER). In response (Request for Additional Information (RAI) No. 1
40 and No. 2, Reference 3), Westinghouse modified the pellet thermal conductivity model.
41 Westinghouse subsequently re-performed the validation cases which were then benchmarked
42 by PNNL to FRAPCON-3 and compared to relevant data. Figures 2.2.1 through 2.2.4 of the
43 supporting TER illustrate these comparisons. Based upon these latest comparisons, PNNL
44 concluded that the revised STAV7.2 UO₂ pellet thermal conductivity model was acceptable.
45 Based upon this assessment, the NRC staff finds the revised UO₂ pellet thermal conductivity
46 model acceptable.

1 Because the nominal fuel density of the in-reactor fuel database used to validate the STAV7.2
2 fuel thermal conductivity model is between 92 percent theoretical density and 97 percent
3 theoretical density, the applicability of STAV7.2 will be limited to within this range.
4

5 Incorporating an empirical uranium-gadolinia thermal conductivity correlation into the
6 FRAPCON-3 model, PNNL compared benchmark cases to the revised STAV7.2 thermal
7 conductivity model (Section 2.2.1 of supporting TER). PNNL concluded that the STAV7.2
8 thermal conductivity adjustment for gadolinia is acceptable and applicable up to the 9.0 wt%
9 gadolinia concentration requested by Westinghouse. Based upon this assessment, the NRC
10 staff finds the revised $\text{UO}_2\text{-Gd}_2\text{O}_3$ pellet thermal conductivity model acceptable and applicable
11 up to 9.0 wt% gadolinia.
12

13 Westinghouse has not requested approval of STAV7.2 for fuel pellets containing additives other
14 than gadolinia, nor has thermal conductivity data for such additives been provided. As such,
15 approval for STAV7.2 will be limited to UO_2 and $\text{UO}_2\text{-Gd}_2\text{O}_3$ fuel pellets with no additives
16 beyond nominal trace elements.
17

18 Gap Conductivity

19 Based upon comments received from PNNL, Westinghouse revised their pellet-cladding gap
20 thermal model (Attachment 1 of Reference 3). The revised model was compared to
21 FRAPCON-3 and found to be acceptably conservative. In addition, the accommodation
22 coefficients for the gap gas species has been updated in STAV7.2. The values of these gas
23 coefficients are identical to values in MATPRO. Based upon this assessment, the NRC staff
24 finds the pellet-cladding gap conductivity model acceptable.
25

26 Fuel Thermal Expansion

27 The fuel thermal expansion model in STAV7.2 has not changed relative to STAV6.2. PNNL
28 conducted a benchmark against FRAPCON-3 (Section 2.4 of supporting TER) and identified
29 that the STAV7.2 code does not model the large increase in fuel volume during fuel melting.
30 Because the code does not model this known phenomenon, its applications will be limited to
31 fuel temperatures less than the melting temperature. Based on this assessment, the NRC staff
32 finds the fuel thermal expansion acceptable with the condition that the use of STAV7.2 is limited
33 to applications where the fuel temperature remains below the melting temperature.
34

35 Fuel Relocation

36 The overall gap thermal conductance model, which is influenced by fuel pellet radial relocation
37 (due to thermal cracking and outward movement), was revised by Westinghouse (Attachment 1
38 of Reference 3) in response to concerns raised by PNNL (Section 2.5 of supporting TER). In
39 conjunction, Westinghouse modified the fuel pellet relocation model. The revised relocation
40 model was benchmarked to FRAPCON-3 and found to be in good agreement (Figures 2.5.1
41 and 2.5.2 of supporting TER). Based upon this assessment, the NRC staff finds the revised
42 fuel pellet relocation model acceptable.

1 Clad-to-Coolant Heat Transfer Model

2 The clad-to-coolant heat transfer model in STAV7.2 is unchanged from STAV6.2. PNNL
3 compared this model to the one used by FRAPCON-3 and concluded that it was acceptable.
4 The clad-to-coolant heat transfer model is not burnup dependent and continues to be
5 applicable.

6
7 Clad Thermal Conductivity

8 Over its range of applicability, the STAV7.2 clad thermal conductivity was compared to the
9 model in FRAPCON-3 (Figure 2.6.2 of supporting TER). PNNL concluded that the STAV7.2
10 oxide conductivity was acceptable. Based upon this assessment, the NRC staff finds the clad
11 thermal conductivity acceptable.

12
13 Clad Oxide Thermal Conductivity

14 Over its range of applicability, the STAV7.2 clad oxide thermal conductivity was compared to
15 the model in FRAPCON-3 (Figure 2.6.1 of supporting TER). PNNL concluded that the
16 STAV7.2 oxide conductivity was acceptable. Based upon this assessment, the NRC staff finds
17 the clad oxide thermal conductivity acceptable.

18
19 Crud Thermal Conductivity

20 PNNL compared the crud conductivity value for BWRs in STAV7.2 against the conductivity
21 assigned in FRAPCON-3 (Section 2.6 of the supporting TER). STAV7.2 utilizes a lower
22 conductivity which will promote a greater temperature drop across the crud layer and
23 conservatively higher fuel and clad temperatures. Based on this, PNNL concluded that the crud
24 conductivity was conservative. The NRC staff agrees with this assessment and finds the crud
25 conductivity value acceptable.

26
27 STAV7.2 Thermal Model Integral Assessment

28 Section 3.2 of WCAP-15836-P (as updated by Attachment 2 of Reference 3) describes the
29 calibration and verification of the STAV7.2 thermal model. The STAV7.2 beginning of life (BOL)
30 fuel centerline temperatures compare well to measured test rods from Halden. In-life fuel
31 temperature data from several Halden test rods were also compared against STAV7.2
32 temperature predictions and found to be in good agreement. As part of the calibration process
33 in STAV7.2, tuning parameters in the code algorithms are adjusted to achieve a best-estimate
34 fit to the empirical database. As such, the values of these tuning parameters become an
35 inherent part of the approved model. The PNNL integral assessment of the thermal models is
36 discussed in Section 2.7 of the supporting TER. Based upon the Halden validation
37 (comparisons of predicted fuel temperature versus the Halden test database) and the PNNL
38 technical assessment, the NRC staff finds the overall interaction of the coupled heat generation
39 and heat transfer models acceptable.

40
41 3.2 Fission Gas Release Model

42
43 The fission gas release model in STAV7.2 is a two-stage diffusion model that simulates the
44 diffusion of gas through the grain to the grain boundary and the release from the grain
45 boundary to the void volume. The diffusion constant in STAV7.2 is modified for the effect of
46 gadolinia. The PNNL technical assessment of the fission gas release model is discussed in
47 Section 3.1 of the supporting TER. In their review, PNNL requested further justification of the
48 diffusion constant for $\text{UO}_2\text{-Gd}_2\text{O}_3$ fuel. In response (RAI No. 14 of Reference 2), Westinghouse
49 referred to empirical data obtained from gamma scans which target certain mobile and

1 immobile fission nuclides. These gamma scans demonstrate the diffusion of fission products
2 (via tracking mobile nuclides) as well as characterize the burnup (via examining the immobile
3 nuclides) for both UO_2 and $\text{UO}_2\text{-Gd}_2\text{O}_3$ fuel. The applicability of the diffusion coefficient and
4 supporting gamma scan data for $\text{UO}_2\text{-Gd}_2\text{O}_3$ fuel up to the requested gadolinia concentration of
5 9.0 wt% was questioned. In response (RAI No. 9 of Reference 4), Westinghouse agreed to
6 modify the diffusion model to saturate the effects of gadolinia at the limit of the empirical
7 database.

8
9 PNNL compared the STAV7.2 fission gas release predictions to FRAPCON-3. While the
10 algorithms and tuning parameters differ, the two codes have reasonable agreement. In
11 most cases, STAV7.2 predicts greater fission gas release which promotes conservative fuel rod
12 internal pressure and fuel temperature calculations.

13 Grain Growth

14 PNNL compared the grain growth model in STAV7.2 against an internationally acknowledged
15 model and empirical data (Section 3.2 of supporting TER). This comparison revealed that
16 STAV7.2 over-predicts grain size which would tend to under-predict fission gas release.

17 STAV7.2 Fission Gas Release Integral Assessment

18
19 Section 3.3 of WCAP-15836-P (as updated by Attachment 2 of Reference 3) describes the
20 calibration and verification of the STAV7.2 fission gas release models. As part of the calibration
21 process in STAV7.2, tuning parameters in both the thermal and athermal code algorithms are
22 adjusted to achieve a best-estimate fit (and in some cases a 95 percent upper-bound) to the
23 empirical database. As such, the value of these tuning parameters become an inherent part of
24 the approved model. The PNNL integral assessment of the fission gas release models is
25 discussed in Section 3.3 of the supporting TER.

26
27
28 The Westinghouse database utilized in the calibration and verification process of the steady-
29 state fission gas release models consisted of 261 BWR rods and 130 pressurized water reactor
30 (PWR) rods. The transient fission gas database consisted of power ramp data from 60 BWR
31 and PWR fuel rods conducted in test reactors (e.g. Studsvik). Appendix C of WCAP-15836-P
32 characterizes the fuel rod database. The $\text{UO}_2\text{-Gd}_2\text{O}_3$ fuel database consists of 15 BWR rods
33 (steady-state) and 3 PWR rods (ramp tested). Although the STAV7.2 code will not be used for
34 PWR fuel rods, the calibration of the model to the expanded PWR database provides a greater
35 degree of certainty to the model, especially at higher burnup.

36
37 Based on a comparison of the STAV7.2 predictions to the measured test data coupled with
38 comparisons with FRAPCON-3, PNNL concluded that the STAV7.2 fission gas release models,
39 with the revised gadolinia diffusion coefficient, is acceptable up to a burnup of 62 GWd/MTU.
40 Although the grain growth model appears non-conservative, its impact on the predicted fission
41 gas release is counteracted by the empirical nature of the calibration process. Based upon the
42 calibration and verification exercise presented in Section 3.3 of WCAP-15836-P (as updated by
43 Attachment 2 of Reference 3) and the PNNL technical assessment, the NRC staff finds the
44 steady-state and transient fission gas release models acceptable up to a peak rod average
45 burnup of 62 GWd/MTU.

3.3 Cladding Corrosion and Crud Models

A model which accurately predicts cladding waterside corrosion and crud deposition is required because the development of a corrosion layer and/or crud layer promotes increased thermal resistance. This increase in thermal resistance may result in higher fuel and cladding temperatures. In addition, the formation of hydrides may result in a reduction of cladding ductility.

The oxidation rate in STAV7.2 for a BWR is a function of both time and linear heat generation rate (LHGR). The overall corrosion rate is the sum of the nodular corrosion rate (athermal) and the diffusion controlled rate (strong function of temperature). Since corrosion rate is temperature dependent, buildup of a crud layer (which promotes higher clad temperatures) will result in an increase in oxidation rate.

Oxidation Rate

PNNL benchmarked STAV7.2 against the BWR oxidation model in FRAPCON-3. An examination of Figure 4.1.1 of the supporting TER reveals that STAV7.2 predicts a significantly larger amount of oxidation for a typical 10x10 BWR fuel design. This conservative oxidation prediction would promote a larger temperature change across the cladding and higher fuel temperatures.

Crud Deposition

PNNL investigated the crud deposition model in STAV7.2. Figure 4.1.2 of the supporting TER illustrates crud buildup as a function of burnup for two operating power histories. Also shown on this figure is the impact of the crud layer on temperature rise across the crud layer. PNNL benchmarked STAV7.2 against the BWR oxidation model in FRAPCON-3 to investigate the effect of crud deposition on oxidation rate. Figure 4.1.3 of the supporting TER illustrates this effect as well as demonstrating that the combined crud and corrosion models in STAV7.2 are conservative relative to FRAPCON-3.

Hydrogen Pickup

The hydriding model is a stand-alone model in STAV7.2 in that it does not affect other calculated quantities. An RAI was issued (RAI No. 22 of Reference 2) requesting further justification of this stand-alone feature and the potential impact of hydriding on clad ductility, especially under accident conditions. In response, Westinghouse stated that STAV7.2 is used to assess the impact on fuel performance during normal operations and AOOs and will not be used to assess the dynamic response during postulated transients such as BWR Control Rod Drop. In response to RAI No. 5 (Reference 4), Westinghouse provided details on the hydrogen pickup model in STAV7.2 (which was not presented in WCAP-15836-P). The PNNL technical assessment of the hydrogen pickup model is presented in Section 4.2 of the supporting TER. Based upon comparisons of the STAV7.2 predicted hydrogen content against (1) the Westinghouse SVEA-96 database, (2) an extensive international database, and (3) FRAPCON-3, PNNL concluded that the hydrogen pickup model in STAV7.2 is acceptable.

STAV7.2 Clad Corrosion and Crud Integral Assessment

Figure 3.6-1 of WCAP-15836-P presents a "typical" application of the STAV7.2 corrosion model to recent Westinghouse BWR cladding oxide measurements. Note that the corrosion and crud deposition models use tuning parameters which may vary for different BWR plants and for different Westinghouse BWR cladding types. Based on the data presented and the PNNL

1 technical assessment, the NRC staff finds the clad oxidation, hydrogen pickup, and crud
2 deposition models capable, for a given database, of predicting clad waterside corrosion and
3 crud deposition. The NRC staff acknowledges that the buildup of corrosion and crud layers
4 may be plant-specific as well as clad-specific. As such, tuning parameters within these
5 corrosion and crud models may need to be adjusted based upon future trends derived from
6 post-irradiation examinations. The application of these models to future licensees will be
7 addressed in WCAP-15942-P.
8

9 3.4 Fuel Densification and Swelling Models

10
11 PNNL compared the pellet volume change due to densification and swelling in STAV7.2 to
12 FRAPCON-3 (Section 5.0 of supporting TER). Examination of Figure 5-1 of the supporting
13 TER reveals that the maximum densification occurs at a higher burnup and the fuel swelling
14 rate is lower in STAV7.2 relative to FRAPCON-3. A less rapid densification will promote lower
15 fuel temperatures at the beginning of life. A lower swelling rate will promote a lower gap
16 conductance and higher fuel temperatures throughout fuel life.
17

18 Each model in a fuel performance code does not need to be independently conservative for
19 every application. It is the interaction of the various models coupled with an appropriate
20 implementation methodology (WCAP-15942-P) which ensures a conservative fuel design.
21 Westinghouse provided a comparison of the STAV7.2 predicted rod void volume to measured
22 data (Figure 3.7-1 of WCAP-15836-P) which shows a best-estimate fit. PNNL concluded that
23 STAV7.2 fuel densification and swelling models are acceptable for thermal analyses and
24 acceptable for evaluating the relative mechanical performance of different fuel designs and
25 operating modes. Based upon the data presented and the PNNL technical assessment, the
26 NRC staff finds the fuel densification and swelling models acceptable. A conservative
27 application of these models will be governed by the mechanical design methodology in
28 WCAP-15942-P.
29

30 3.5 Fuel Rod Mechanical Properties

31
32 The modeling of mechanical fuel rod behavior in STAV7.2 assumes a rigid pellet (i.e. no fuel
33 creep) and the fuel strain (i.e., fission product swelling, densification, thermal expansion,
34 relocation, and rearrangement) determines the amount of elastic-plastic strain in the cladding
35 when contact between the fuel and the cladding is achieved. The cladding mechanical
36 properties are modeled in STAV7.2 in order to predict the cladding response to in-reactor
37 stresses and predict elastic and plastic deformation and cladding creep.
38

39 Since mechanical properties are strongly dependent on cladding types (i.e. alloying and heat
40 treatments), the NRC staff and Westinghouse agreed early in the review to limit the applicability
41 of STAV7.2 to fully recrystallized (RXA) Zircaloy-2 (RAI No. 25, Reference 2).
42

43 Most of the mechanical properties in STAV7.2 are identical to those in the previously approved
44 STAV6.2. Those models that have changed are cladding creep, Young's modulus, and yield
45 strength.

1 Irradiation Creep

2 Section 3.4 of WCAP-15836-P (as updated by Attachment 2 of Reference 3) describes the
3 calibration and verification of the STAV7.2 cladding creep model. The database used to
4 calibrate the creep model is primarily in-reactor cladding creep data obtained from tests
5 performed by Babcock & Wilcox in the Oconee nuclear power plant under an Electric Power
6 Research Institute (EPRI) program. Due to the empirical nature of the calibration process, the
7 values of the tuning parameters become inherently part of the approved model. PNNL
8 evaluated the calibration of the creep model (Section 6.2.1 of supporting TER) and concluded
9 that the creep model fit the Oconee data in a best-estimate manner.

10
11 Westinghouse's verification of the creep model (RXA Zirc-2) consisted of comparisons to
12 measured creep down from 24 BWR commercial fuel rods and 8 experimental fuel rods
13 irradiated in the Studsvik test reactor. Comparisons presented in Section 3.4.2 of
14 WCAP-15836-P were supplemented with additional creep data in response to RAI No. 21
15 (Reference 2). Since most of this creep data were based on an earlier version of
16 Westinghouse BWR cladding, an RAI was issued requesting further justification of the creep
17 model to the latest RXA Zirc-2 cladding alloys (e.g., LK2, LK2+, and LK3). In response (RAI
18 No. 3 of Reference 4), Westinghouse provided creep data which showed little difference in
19 creep behavior between the different alloys.

20
21 PNNL benchmarked the STAV7.2 creep model against FRAPCON-3. Based upon the
22 verification database and comparisons to FRAPCON-3, PNNL concluded that the creep
23 correlation in STAV7.2 is acceptable for fully RXA Zircaloy-2 cladding. Based upon the
24 calibration and verification exercise presented in Section 3.4 of WCAP-15836-P (and in
25 response to the RAIs) and the PNNL technical assessment, the NRC staff finds the creep
26 model in STAV7.2 acceptable.

27
28 High Stress Creep Model

29 PNNL benchmarked the steady-state creep model in STAV7.2 against the FRAPCON-3 model
30 at BWR conditions and high stress (Section 6.2.2 of supporting TER). Examination of figure
31 6.2.3 of the supporting TER reveals that the two models predict similar values for creep rate
32 with STAV7.2 predictions slightly more conservative. The Westinghouse database used to
33 verify the creep model was limited to a hoop stress of 100 MPa. PNNL considered whether the
34 model's verification to a limited database would necessitate a code limitation. PNNL concluded
35 that since the application of STAV7.2 will be limited by the no-clad-lift-off criteria (which at
36 approximately 75 MPa for a typical BWR fuel rod is within the 100 MPa database), the creep
37 model is acceptable without a specified limitation due to its supporting database.

38
39 Yield Strength and Young's Modulus

40 STAV7.2 uses Hooke's Law to relate stress and strain in the elastic region and a modified
41 power law to relate stress and strain in the plastic region. PNNL compared the coefficients in
42 STAV7.2 against FRAPCON-3 and mechanical test data and concluded that STAV7.2 strength
43 coefficients were excessively high for irradiated cladding. In response to RAI No. 8
44 (Reference 4), Westinghouse provided further justification for the strength coefficients in
45 STAV7.2. As part of their response, Westinghouse agreed to better treat the strength
46 coefficient at high temperatures. The STAV7.2 yield strength correlation was modified to
47 include a correction factor, which is a function of the liner thickness.
48 The yield stress used in STAV7.2 will be the larger of either the STAV7.2 predicted yield stress
49 (adjusted by correction factor) or the yield stress predicted by the original STAV6.2 model.

1 Because the STAV6.2 correlation does not contain a term to correct for the thickness of a clad
2 liner, applicability of STAV7.2 will be limited to fuel rod designs with a maximum clad liner
3 thickness of 4 mils (nominal) which is the upper extent of the database used in the calibration
4 and verification of the yield stress model.
5

6 PNNL compared the revised STAV7.2 yield stress model to FRAPCON-3 for both unirradiated
7 and irradiated cladding. Examination of Figure 6.2.4 through 6.2.6 reveal that the behavior of
8 both models (with fluence and temperature) are similar in shape and magnitude. Neither PNNL
9 nor Westinghouse was able to produce data showing yield stress above 698K for RXA Zirc-2
10 cladding. As a result of this lack of data within the industry, the applicability of STAV7.2 will be
11 limited to applications with cladding average temperature at any axial node less than 698K.
12 Based on the data presented in WCAP-15836-P and in response to RAIs and the comparison
13 to FRAPCON-3, PNNL concluded that the yield stress model is acceptable.
14

15 Up to the applicability limit of 698K, the model used for Young's modulus in STAV7.2 is identical
16 to the model in FRAPCON-3. PNNL concluded that this is acceptable.
17

18 Based upon the data presented in WCAP-15836-P and in response to RAIs and the PNNL
19 technical assessment, the NRC staff finds the revised yield stress model acceptable up to a
20 clad average temperature of 698K (425EC) at any axial node and a clad liner thickness of 4 mils
21 (nominal).
22

23 Pellet-Cladding Mechanical Interaction

24 Section 3.8.2 of WCAP-15836-P (as updated by Attachment 2 of Reference 3) describes the
25 verification of the STAV7.2 PCMI model. PNNL assessed the pellet-cladding mechanical
26 interaction in STAV7.2 against Halden test IFA-404.1. This Halden rod was ramped up from
27 zero power to approximately 50 kW/m and then back down to zero power. Instrumentation
28 measured axial and hoop strains during the power ramps. The STAV7.2 prediction of hoop
29 strain was compared to the Halden data and found to be in reasonable agreement. PNNL
30 concluded that the STAV7.2 predictions of hoop and axial strain for this case are acceptable.
31 Based on the PNNL technical assessment, the NRC staff finds the PCMI modeling in STAV7.2
32 acceptable.
33

34 3.6 Void Volume

35
36 The void volume in STAV7.2 is calculated in the same manner as in STAV6.2. The predicted
37 void volume is affected by many of the models in STAV7.2, including fuel pellet swelling,
38 densification, cladding creep, fuel rod growth, fission gas release, and temperature distribution.
39

40 Fuel Rod Growth

41 The model for fuel rod growth in STAV7.2 for BWR cladding is the same as the model in
42 STAV6.2. PNNL compared the PWR and BWR fuel growth models in STAV7.2 against
43 FRAPCON-3 for Zircaloy-2 (BWR) and Zircaloy-4 (PWR). Note that the FRAPCON-3 models
44 are based on an EPRI model and has been validated to a local burnup of 65 GWd/MTU.
45 Examination of Figure 7.1.1 in the supporting TER reveals good agreement between the
46 models in STAV7.2 and FRAPCON-3. The Westinghouse database employed to validate
47 STAV7.2 consists of BWR fuel data beyond 62 GWd/MTU. Based on the data comparisons,
48 the PNNL technical assessment concluded that the STAV7.2 model for BWR fuel rod growth is
49 acceptable. Based upon the data presented in WCAP-15836-P (and Figure 4.2-7 of

1 WCAP-15942-P, currently under review) and the PNNL technical assessment, the NRC staff
2 finds the STAV7.2 model for BWR fuel rod growth acceptable up to 62 GWd/MTU.

3
4 Plenum Gas Temperature

5 The plenum gas temperature in STAV7.2 is set equal to the coolant temperature at the axial
6 node of the plenum. PNNL recognized that this assumption is slightly non-conservative
7 because some heat will be transferred into the plenum from the end pellet and some heat will
8 be produced in the plenum by gamma heating in the plenum spring (Section 7.2 of supporting
9 TER). In response to RAI No. 6 (Reference 4), Westinghouse agrees that this simplistic
10 modeling assumption is non-conservative, especially for the plenum in the part length rod.
11 However, Westinghouse stated that “this effect on hot gas pressure is within the noise of the
12 ability to calculate pressure and insignificant relative to the uncertainties that are taken into
13 account.” PNNL agrees with these assessments and concluded that the effects of this slight
14 non-conservatism will be minor. The NRC staff agrees with the PNNL technical assessment of
15 the plenum gas modeling in STAV7.2 and finds it acceptable.

16
17 Void Volume Integral Assessment

18 Section 3.7 of WCAP-15836-P (as updated by Attachment 2 of Reference 3) describes the
19 verification of the STAV7.2 void volume calculations. The PNNL integral assessment of the
20 void volume calculations is discussed in Section 7.0 of the supporting TER.

21
22 The Westinghouse database, utilized in the verification process, is a subset of the database
23 used in the calibration and verification of the steady-state fission gas release models. Of this
24 database, hot cell void volume measurements were made on 66 BWR and 129 PWR fuel rods.
25 Figure 3.7.1 (as updated by Attachment 2 of Reference 3) illustrates measured versus
26 predicted void volumes at room temperature. The BWR and PWR database supporting the
27 void volume calculations has a maximum burnup of 52 GWd/MTU and 61 GWd/MTU,
28 respectively.

29
30 Based on the data comparisons, the PNNL technical assessment states that the STAV7.2 code
31 predicts void volume in a best-estimate manner with a relatively small degree of scatter. PNNL
32 concluded that the STAV7.2 void volume model is acceptable. Based upon the data presented
33 in WCAP-15836-P and the PNNL technical assessment, the NRC staff finds the void volume
34 calculation acceptable.

35
36 3.7 Licensing Applications

37
38 In order to investigate the synergistic interaction of coupled models and ensure a conservative
39 licensing application of STAV7.2, Westinghouse provided sample fuel rod design applications
40 of a typical BWR fuel assembly and PNNL benchmarked these licensing applications to
41 FRAPCON-3. The fuel mechanical design methodology, including the application of
42 uncertainties, will be addressed in WCAP-15942-P. The scope of this investigation is to assess
43 the ability of STAV7.2 to predict fuel rod performance in a best estimate or conservative
44 manner depending on the analyses.

1 Fuel Melting

2 PNNL benchmarked the fuel melting temperature in STAV7.2 against FRAPCON-3 (Section 8.1
3 of the supporting TER). In addition to a comparison of steady-state fuel melting temperature
4 with burnup, fuel temperature benchmark cases, including six segmented power histories
5 (SPHs) and an anticipated operational occurrence (AOO) for full length and part length UO₂
6 rods, were performed. PNNL concluded that for the fuel melt analysis, STAV7.2 predicts
7 conservative values for fuel melting temperature and conservative values for maximum fuel
8 temperature for a rod average burnup up to 62 GWd/MTU. Based upon the PNNL technical
9 assessment, the NRC staff finds the application of STAV7.2 to the fuel melting analysis
10 acceptable.

11
12 Fuel Stored Energy

13 PNNL benchmarked the peak node centerline temperature in STAV7.2 against FRAPCON-3 for
14 two different power histories (Section 8.2 of supporting TER). For both cases, STAV7.2
15 predictions of peak node centerline temperature were conservatively higher than FRAPCON-3.
16 PNNL concluded that STAV7.2 is acceptable for LOCA initialization (input to CHACHA-3) at
17 reasonable power level and temperature, but may be excessively conservative at high power
18 levels and temperatures due to conservative fission gas release predictions. Based upon the
19 PNNL technical assessment, the NRC staff finds the application of STAV7.2 to the LOCA
20 stored energy analysis (via input to CHACHA-3) acceptable.

21
22 Fuel Rod Internal Pressure

23 PNNL benchmarked the rod internal pressure in STAV7.2 against FRAPCON-3 over a range of
24 power levels including the six SPH and several different AOO power pulse scenarios for full
25 length and part length fuel rods (Section 8.3 of supporting TER). In each case, STAV7.2
26 predicted conservatively higher rod internal pressures than FRAPCON-3. PNNL found that the
27 difference between the two codes' prediction of rod internal pressure is due to the
28 conservatively high fission gas release predicted by STAV7.2. PNNL concluded that the
29 STAV7.2 code is acceptable for application to fuel rod pressure analysis. Based on the PNNL
30 technical assessment, the NRC staff finds the application of STAV7.2 to the rod internal
31 pressure analysis acceptable.

32
33 Clad Strain

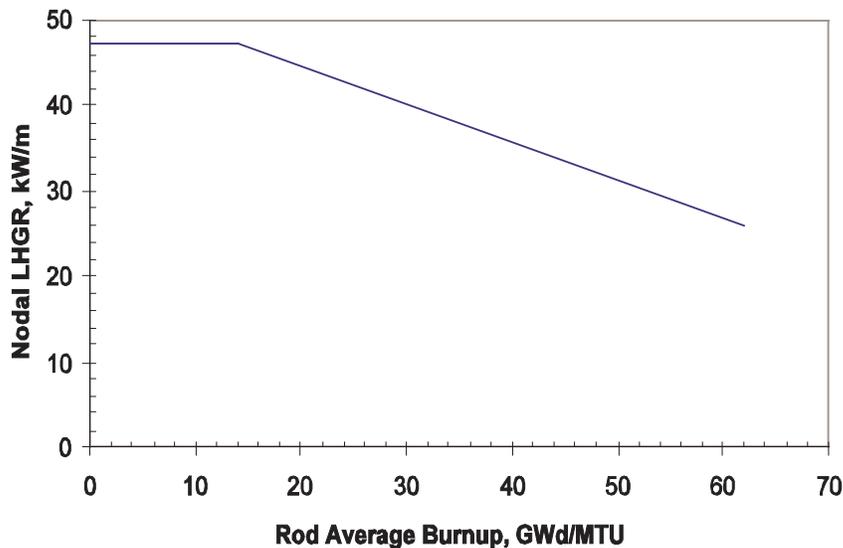
34 PNNL benchmarked the maximum cladding strain in STAV7.2 against FRAPCON-3 for a
35 limiting AOO case including three power pulses each lasting thirty hours (Section 8.4 of
36 supporting TER). The maximum hoop strain predicted by STAV7.2 was greater than that
37 predicted by FRAPCON-3. PNNL determined that the reason FRAPCON-3 predicts less strain
38 than STAV7.2 is because no stress is transferred to the cladding while the relocation of the
39 pellet is being taken up in FRAPCON-3. While in STAV7.2, a condition of soft contact is
40 defined where some stress is transferred to the cladding while the relocation is being taken up.
41 PNNL concluded that the STAV7.2 code is acceptable for application to cladding strain
42 analyses. Based on the PNNL technical assessment, the NRC staff finds the application of
43 STAV7.2 to cladding strain analyses acceptable.

44
45 3.8 STAV7.2 LHGR Limitation

46
47 Longer cycle lengths and higher power core reload designs (e.g. Extended Power Uprate)
48 promote more aggressive fuel utilization with higher fuel rod power throughout fuel rod burnup.
49 It is important to verify the applicability of the STAV7.2 fuel performance models (and

1 supporting fuel experience database) to these higher power cores. In response to an RAI
2 (Reference 5), Westinghouse provided a proposed LHGR limit as a function of rod average
3 burnup along with the power histories for the fuel rods in their fission gas release and fuel
4 temperature database. Figures 3 through 6 of Reference 5 provide an investigation of the
5 STAV7.2 model predictions versus measurements taken from fuel rods with power histories
6 adjacent to the proposed LHGR limit. In Section 8.5 of the supporting TER, PNNL concluded
7 that the calibration and verification data used in the development of STAV7.2 support operation
8 at or below the LHGR limit shown below. It should be noted that this limit applies only to
9 steady-state LHGR and does not apply to transient LHGR such as for an AOO. The peak
10 LHGR during a transient may exceed this LHGR limit for the short duration of the transient but
11 must meet the LHGR versus time duration used for analyzing AOO events. Based upon the
12 information presented in Reference 5 along with the PNNL technical assessment, the NRC staff
13 finds that STAV7.2 is applicable up to the nodal power profile depicted below.
14

15 The figure below illustrates the peak LHGR ≤ 47.3 kW/m from BOL to a rod average burnup of
16 14 GWd/MTU, then linearly decreasing to 26 kW/m at a rod average burnup of 62 GWd/MTU.



1 3.9 VIK-3
2

3 The computer code VIK-3, described in Section 4 of WCAP-15836-P, performs cladding stress
4 and end plug weld area stress analyses. A majority of the code features remain unchanged
5 from VIK-2, approved in CENPD-285-P-A (Reference 6). The PNNL technical assessment of
6 VIK-3 is provided in Section 9 of the supporting TER.
7

8 In VIK-3, stress calculations can be performed as a function of fuel rod burnup using STAV7.2
9 material properties, fuel rod parameter inputs, and loads. This code feature allows the code a
10 greater degree of integration with STAV7.2.
11

12 The Paidoussis correlation has been added to VIK-3 which provides amplitudes due to flow
13 induced forces to the rod bending calculation. In response to RAI No. 1 (Reference 2) and
14 RAI No. 7 (Reference 4), Westinghouse provided a comparison of the VIK-3 predictions against
15 the experimental data used to validate the code.
16

17 The finite difference technique used in VIK-2 to calculate the stress and temperature
18 distribution in the bottom end plug has been replaced with a finite element technique in VIK-3.
19 In response to RAI No.3 (Reference 2), Westinghouse provided a more detailed description of
20 the finite element code and a typical finite element mesh.
21

22 Based upon their review of the information presented in WCAP-15836-P and in response to
23 RAIs, PNNL concluded that the VIK-3 code is acceptable for cladding stress analysis and end
24 plug weld area stress analyses. Based upon the PNNL technical assessment, the NRC staff
25 finds the VIK-3 code acceptable.
26

27 3.10 COLLAPS-3.3D
28

29 The computer code COLLAPS-3.3D, described in Section 6 of WCAP-15836-P, calculates
30 cladding ovalization due to creep, up to the point of mechanical instability and creep collapse.
31 A majority of the code features remain unchanged from COLLAPS-3.2S, approved in
32 CENPD-285-P-A (Reference 6). The PNNL technical assessment of COLLAPS-3.3D is
33 provided in Section 10 of the supporting TER.
34

35 Changes incorporated into COLLAPS-3.3D are listed below:
36

- 37 • Use of double precision to increase the computational accuracy of the code.
- 38
- 39 • Optional correction to the infinitely long solution to account for the effect of the pellet
40 support provided at the ends of a finite length pellet-to-pellet axial gap.
- 41
- 42 • STAV7.2 creep correlation for fully-annealed cladding.
43

44 In response to RAI No. 1 and No. 2 (Reference 2), Westinghouse provided background
45 information on the Studsvik data used to validate the creep model. In response to RAI No. 3
46 (Reference 2), Westinghouse provided background information on the collapse data used to
47 verify the COLLAPS-3.3D model. In response to RAI No. 4 (Reference 2), Westinghouse
48 provided further discussion on the determination of the finite gap length.
49

1 Based upon the information presented in WCAP-15836 and in response to RAIs, PNNL
2 concluded that the COLLAPS-3.3D code is acceptable for calculating cladding ovality due to
3 creep, up to the point of mechanical instability and creep collapse. Based upon the PNNL
4 technical assessment, the NRC staff finds the COLLAPS-3.3D code acceptable.
5

6 4.0 CONCLUSION

7
8 Based upon its review of this TR and technical support provided by the PNNL, the NRC staff
9 finds Westinghouse's suite of BWR fuel performance computer models, STAV7.2, VIK-3, and
10 COLLAPS II Version 3.3D, acceptable. Licensees referencing this TR will need to comply with
11 the conditions and limitations listed in Section 5.0.
12

13 5.0 CONDITIONS AND LIMITATIONS

14
15 Licensees referencing WCAP-15836-P must ensure compliance with the following conditions
16 and limitations:
17

- 18 1. STAV7.2 is approved for modeling BWR fuel rods with the following limitations.
19
 - 20 a. Solid UO₂ fuel pellet with a maximum gadolinia content of 9.0 wt%.
21 [Requested by Westinghouse, see Section 3.1]
 - 22 b. No substance beyond gadolinia and nominal trace elements shall be added to the
23 fuel pellet for the purposes of altering its physical characteristics.
24 [Fuel additives not part of review scope, see Section 3.1]
 - 25 c. Nominal fuel pellet density between 92 - 97 percent theoretical.
26 [Extent of fuel thermal conductivity database, see Section 3.1]
 - 27 d. Fully RXA Zircaloy-2 fuel clad material.
28 [Clad properties calibrated to RXA Zircaloy-2 database, see Section 3.5]
 - 29 e. For fuel rods with a clad liner (e.g. natural zirconium), the liner thickness shall be
30 no greater than 4 mils (nominal).
31 [Extent of cladding yield stress database, see Section 3.5]
 - 32 f. Peak rod average burnup limit of 62 GWd/MTU.
33 [Requested by Westinghouse, see Section 1.0]
- 34
35 2. STAV7.2 shall not be used to model fuel above incipient fuel melting temperatures.
36 [Limitation on fuel thermal expansion model, see Section 3.1]
37
- 38 3. STAV7.2 shall not be used to model fuel rods with an average cladding temperature
39 above 698K (425 EC) at any axial node.
40 [Extent of cladding yield stress database, see Section 3.5]
41
- 42 4. STAV7.2 shall be used only within the range for which fuel performance data were
43 acceptable or for which the verifications discussed in WCAP-15836-P and responses to
44 RAIs were performed. For example, Section 3.8 describes a LGHR limit based upon the
45 calibration and verification database of STAV7.2.
46 [Applicability of STAV7.2 shall remain within NRC scope of review and acceptance, see
47 Section 3]
48

- 1 5. Due to the empirical nature of the STAV7.2 calibration and validation process, the
2 specific values of the equation constants and tuning parameters derived in
3 WCAP-15836-P (as updated by RAIs, e.g. Attachment 2 of Reference 3) become
4 inherently part of the approved models. Thus, these values may not be updated without
5 further NRC review. Exceptions include the BWR cladding corrosion constants
6 (Table 2.2.51), crud deposition constants (Table 2.2.5-2), and rod nodal power
7 uncertainties for the BWR "Older" data (Uncontrolled and Controlled Cells in Table 3.3-
8 1). These exceptions will be addressed as part of the implementation methodology in
9 WCAP-15942-P.

10 [Applicability of STAV7.2 shall remain within NRC scope of review and acceptance, see
11 Section 3]
12

13 6.0 REFERENCES

- 14
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43

44 Principal Contributor: Paul Clifford, NRR

45
46 Date: June 2, 2005